UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30303-1257

February 24, 2011

Mr. Michael J. Annacone
Vice President
Brunswick Steam Electric Plant
P.O. Box 10429
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SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT - NRC COMPONENT DESIGN BASES INSPECTION - INSPECTION REPORT 05000325/2010008 AND 05000324/2010008

Dear Mr. Annacone:

On January 21, 2011, U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your Brunswick Steam Electric Plant. The enclosed inspection report documents the inspection results, which were discussed on February 3, 2011, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your licenses. The team reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents three NRC-identified findings of very low safety significance (Green), which were determined to involve violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent the NRC Enforcement Policy. If you contest these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Hatch. Further, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Resident Inspector at Brunswick. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Binoy Desai, Chief
Engineering Branch 1
Division of Reactor Safety

Enclosure: Inspection Report 05000324/2010008 and 325/2010008,
w/Attachment: Supplemental Information

Docket No.: 50-325, 50-324
License No.: DPR-71, DPR-62

cc w/encl: (See page 3)

X☐ PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE X☐ NON-SENSITIVE

ADAMS: X ☐ Yes ACCESSION NUMBER:_ML110550744_______________ xBD 2/24/11☐SUNSI REVIEW COMPLETE

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Letter to Michael J. Annacone from Binoy Desai dated February 24, 2011.

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT - NRC COMPONENT DESIGN BASES INSPECTION - INSPECTION REPORT 05000325/2010008 AND 05000324/2010008; PRELIMINARY GREATER THAN GREEN FINDING

Distribution w/encl:
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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.: 50-325, 50-324

License Nos.: DPR-72 & 61

Report Nos.: 05000324/2010008 AND 05000325/2010008

Licensee: Carolina Power & Light Corporation

Facility: Brunswick Steam Electric Plant, Units 1 & 2

Location: Southport, NC

Dates: October 4 – January 21, 2011

Inspectors: D. Jones, Senior Reactor Inspector (Lead)
A. Alen, Reactor Inspector
D. Mas Penaranda, Reactor Inspector
T. Lighty, Project Engineer
G. Ottenberg, Resident Inspector, Oconee
H. Campbell, Contractor
J. Nicely, Contractor

Approved by: Binoy B. Desai, Chief
Engineering Branch 1
Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000324/2010008 and 05000325/2010008; 10/04/2010 – 01/21/2011; Brunswick Steam Electric Plant; Component Design Bases Inspection.

This inspection was conducted by a team of five NRC inspectors from the Region II office, and two NRC contract inspectors. Three Green non-cited violations (NCV) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using the NRC Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” (ROP) Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to use conservative motor control center (MCC) voltage inputs when calculating motor actuator output torque and control circuit voltages for safety-related motor operator valve (MOV) motors that would be required to operate during design bases events. Specifically, the licensee used steady state MCC voltages instead of more limiting transient voltages that would occur during design bases load sequencing. The licensee entered these issues into their corrective action program as NCRs 427745 and 429541and performed additional analyses to demonstrate operability of the MOVs.

  The licensee’s failure to evaluate MOV motor actuator output torque using transient MCC voltages, and the failure to evaluate whether those MOVs would have adequate control power voltage was a performance deficiency. The performance deficiency was more that minor because it is associated with the mitigating systems cornerstone attribute of design control and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the use of non-conservative voltage inputs for MOV calculations could result in the failure of the components to perform their design bases functions during an event. The inspectors conducted a Phase 1 SDP in accordance with IMC 0609.04, “Initial Screening and Characterization of Findings,” and determined the finding to be of very low safety significance (Green) because it was a design deficiency confirmed not to result in the loss of operability or functionality, did not represent the loss of a system function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding directly involved the cross-cutting aspect of procedural compliance and personnel follow procedures within the Work Practices component of the Human Resources area [H.4(b)]. [Section 1R21.2.5]

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to assure that conditions adverse to quality, such as deficiencies, were promptly identified and corrected. Specifically, after identifying that the Unit 1 and 2 isolation override switches associated with the hardened wet well (Torus) vents should have been scoped in the maintenance rule, the licensee failed to ensure the circuitry was monitored for functionality. Because the circuitry was not monitored, a relay in the Unit 1 circuitry degraded unacceptably without the licensee's knowledge. This finding does not present an immediate safety concern because as an immediate corrective action the failed relay in the Unit 1 control circuitry was replaced. The licensee entered the issue into their corrective action program as NCR 428054.

  The licensee's failure to identify that the isolation override control switches were not being tested in the manner that they would be operated in the EOPs was a performance deficiency. The performance deficiency was more than minor because it is associated with the mitigating systems cornerstone attribute of equipment performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, not testing the isolation override circuitry resulted in a failed component going undetected that adversely impacted the ability to mitigate an event with the hardened wet well vent.

  Using Manual Chapter Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the inspectors determined the finding required a Phase 2 analysis because the finding represented an actual loss of safety function of non-Technical Specification equipment designated as risk significant for greater than 24 hours. The Phase 2 analysis determined that this finding was potentially greater than green; therefore, a Phase 3 analysis was performed by a regional senior reactor analyst (SRA). The SRA determined the change in risk through use of the plant specific risk model. The function of the wet well vents was modeled, but the model did not include the ability to supply air manually to the air operated valve as a backup to the control room switches. A human reliability analysis was performed, and model adjustments were made so the performance deficiency's impact could be analyzed, given the backup method was available. This backup method's availability resulted in the findings risk increase to be low enough to be considered a Green SDP item. A cross-cutting aspect was not identified because the finding does not represent current performance. [Section 1R21.2.13]

- Green. The team identified a non-cited violation of Technical Specification (TS) 5.4.1, Procedures, for the licensee's failure to maintain adequate abnormal operating procedures (AOP) for opening a service water cross-tie valve during a loss of offsite power (LOOP) event. The valve would not open against system differential pressure (dp) and the licensee's corrective actions did not address the valve's manipulation in AOPs. The licensee entered the issue into their corrective action program as NCR 428809.

After discovering the difficulty of opening the service water cross-tie valves against a maximum differential pressure, the licensee's failure to provide appropriate procedural guidance to assure the operation of the valves during a LOOP event was a performance deficiency. The performance deficiency was more that minor because it is associated with the mitigating systems cornerstone attribute of equipment performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences because the SW-V146 valve would not open against a system pressure of 75 psid; and, if this condition was left uncorrected, the ability to complete required operator actions in procedures 0AOP18.0 and 0AOP36.1 during a LOOP would be adversely affected. Using Manual Chapter Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the inspectors determined the finding required a Phase 2 analysis because the finding represented an actual loss of safety function of non-Technical Specification equipment designated as risk significant for greater than 24 hours. The Phase 2 analysis determined that this finding was potentially greater than green; therefore, a Phase 3 analysis was performed by a regional senior reactor analyst (SRA). The SRA determined the combined risk associated with the valve's function to provide alternate flow to safety-related heat exchangers was very low. In addition, there was a good chance of operator recovery due to the long time period that was available before manipulation of the valve was required. These factors resulted in a risk value corresponding to a Green finding. The finding directly involved the cross-cutting aspect of thoroughness of evaluation within the Corrective Action Program component of the Problem Identification and Resolution area [P.1(c)]. [Section 1R21.3]

B. Licensee-Identified Violations

None

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Inspection Sample Selection Process

The team selected risk significant components and operator actions for review using information contained in the licensee's Probabilistic Risk Assessment (PRA). In general, this included components and operator actions that had a risk achievement worth factor greater than 1.3 or Birnbaum value greater than 1 $X10^{-6}$. The sample included seventeen components, three operator actions, and four operating experience (OE) items.

The team performed a margin assessment and a detailed review of the selected risk-significant components to verify that the design bases had been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for a detailed review. These reliability issues included items related to failed performance test results, significant corrective action, repeated maintenance, maintenance rule status, Regulatory Issue Summary (RIS) 05-020 (formerly Generic Letter (GL) 91-18) conditions, NRC resident inspector input of problem equipment, System Health Reports, industry operating experience, and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense-in-depth margins. An overall summary of the reviews performed and the specific inspection findings identified is included in the following sections of the report.

.2 Results of Detailed Reviews

.2.1 Condensate Storage Tank (CST) Level Switches

a. Inspection Scope

The team reviewed Units 1 and 2 system flow diagrams, instrumentation and control (I&C) drawings, elementary and schematic diagrams, instrument setpoint and uncertainty calculations, as well as calibration procedures and calibration test records to verify that the I&C for the CST level switches were in accordance with design bases documents. Also, the team reviewed engineering change (EC) 66710 to verify proper replacement of the CST level switches. The last two completed calibration test records were reviewed to confirm that instrument setpoints were consistent with setpoint calculations. The team also performed field walkdowns of the CST level switches to observe the existing conditions and configurations. In addition, the team reviewed a sample of condition reports to confirm that the licensee adequately identified and corrected adverse conditions.

b. Findings

No findings were identified.

.2.2 Reactor Core Isolation Cooling (RCIC) Low Pressure Switch (Suction Pressure Trip Time Delay)

a. Inspection Scope

The team reviewed the Units 1 and 2 system flow diagrams, I&C drawings, elementary and schematic diagrams, instrument setpoint and uncertainty calculations, as well as calibration procedures and test records to verify that the instrumentation and controls for the RCIC low pressure switches were in accordance with design bases documents. The team also reviewed EC 67305 to verify proper installation of RCIC Pump Trip Time Delay Relay. For the RCIC low pressure switch, the team reviewed documentation of completed surveillance tests to verify that equipment performance was appropriately monitored and maintained consistent with the design and licensing basis. The team performed field walkdowns of the switches to observe the existing conditions. In addition, the team reviewed a sample of condition reports to confirm that the licensee adequately identified and corrected adverse conditions.

b. Findings

No findings were identified.

.2.3 Shutdown Cooling Inboard Suction Valve (E11-F009)

a. Inspection Scope

The team reviewed the Units 1 and 2 residual heat removal (RHR) shutdown cooling inboard suction valves, E11-F009 to verify they were capable of performing their design bases function. The team reviewed the licensee's calculations of operational margin and verified important inputs into the calculations were sufficiently conservative. The team also verified that the in-field setup of switch settings for the valve actuators were within the setup window assumed in design margin calculations, and verified that test equipment accuracies were considered. Modifications to the actuators were reviewed to verify the adequacy of the design changes. The maintenance history of the valves and actuators and system health reports were also reviewed to examine mechanical condition of the components. Local leak rate testing results were reviewed to verify that the valves were meeting containment isolation leakage assumptions. The team reviewed calculations for degraded voltage at the motor operated valve (MOV) terminals to ensure that worst-case voltage was used in calculating available motor output torque when determining margin. The team reviewed calculations that establish control circuit voltage drop, short circuit, and protection/coordination including thermal overload sizing and testing to verify the capability of the valve to operate during design bases events.

b. Findings

No findings were identified.

.2.4 2C Conventional Service Water (CSW) Pump

a. Inspection Scope

The team reviewed the Final Safety Analysis Report (FSAR), Technical Specification (TS), design bases documents (DBD), applicable plant calculations, and drawings to identify the design bases requirements of the CSW pumps. The team examined system health reports, records of surveillance testing and maintenance activities, and applicable corrective actions to verify that potential degradation was being monitored, prevented and/or corrected. After discussion with the system engineers, the team performed a walkdown of the pump area to examine the visible material condition of the pump, and to verify that the installation was consistent with design documentation. The team reviewed hydraulic calculations to verify that the potential pump degradation assumed in the in-service testing (IST) surveillances would not prevent pump from performing its safety-related function and that design flow and pressure requirements were correctly translated into IST acceptance criteria. The team reviewed documentation that identified features of the system differential pressure flow elements, to ensure the original design requirements had been properly incorporated into plant calibration procedures. The team also reviewed the hydraulic calculations to verify that runout flow, net positive suction head (NPSH) and vortex calculations for the pump were acceptable. The team reviewed calculations that establish voltage drop, protection and coordination, motor brake horse power (BHP) requirements, and short circuit for the motor power supply and feeder cable to verify that adequate voltage was available to operate the CSW pump motor.

b. Findings

No findings were identified.

.2.5 RHR Torus Discharge Isolation Valve (2-E11-F028A/B)

a. Inspection Scope

The team reviewed the FSAR, TS, applicable plant calculations, and drawings to identify the design bases requirements of valves 2-E11-F028A and 2-E11-F028B. The team reviewed engineering changes and associated valve modifications to determine the impact of the changes on the valve's function. The team examined vendor documentation, system health reports, maintenance rule scoping and failure information, records of surveillance testing, maintenance activities, and applicable corrective actions to verify that potential degradation was being monitored and prevented or corrected. The team also performed several interviews with plant personnel to discuss the history of valve testing, maintenance, and details of the corrective actions that had been completed. The team also verified that the highest differential pressure was used to determine the maximum valve opening and/or closing requirements to ensure that the valve would perform its intended safety-related design basis function. The team reviewed testing procedures and diagnostic valve test results to verify the MOV was tested in a manner that would detect a malfunctioning valve. The team reviewed vendor recommendations for preventative maintenance and operation to verify that the maintenance practices ensure that design bases requirements are continually met. The team also conducted a field walkdown of the valves to verify that the installed configuration is consistent with the design bases and plant drawings. The team reviewed calculations for degraded voltage at the MOV terminals to ensure worst-case voltage was used in calculating available motor output torque when determining margin. The team reviewed calculations that establish control circuit voltage drop, short circuit, and protection/coordination including thermal overload sizing and testing to verify the capability of the valve to operate during design bases events.

b. Findings

Introduction: An NRC identified Green non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified for the licensee's failure to use conservative motor control center (MCC) voltage inputs when calculating motor actuator output torque and control circuit voltages for safety-related motor operated valve (MOV) motors that would be required to operate during design bases events. Specifically, the licensee used steady state MCC voltages instead of more limiting transient voltages that would occur during design bases load sequencing. There were approximately 90 valves that were affected.

Description: The team determined that the licensee did not use transient MCC voltages that would occur during design bases load sequencing to evaluate MOV motor actuator output torque. Further, the licensee did not evaluate whether these MOVs would have adequate control power voltage to actuate when they were required to operate.

Brunswick's mechanical analysis and calculations for safety-related MOVs are performed in accordance with Nuclear Generation Group Standard Procedure EGR-NGGC-0203, "MOV Performance Prediction, Actuator Settings, & Diagnostic Test Data Reconciliation," Rev 15, which states that minimum available motor torque (MAMTR) is calculated in accordance with Procedure EGR-NGGC-0101 "Electrical Calculation of Motor Output Torque for AC/DC MOVs," Rev 10. Procedure EGR-NGGC-0101, states: "For calculation of MAMTR, the minimum available MCC source voltage for the scenario under consideration must be used. As a first approach, the Minimum Transient Criteria Voltage for the applicable source should be used. Depending on the specific operating scenario/power source, it may also be possible to justify the use of other voltages in calculating minimum available motor output torque. A justification for the use of minimum MCC source voltages not based on the Minimum Transient Criteria Voltage must be provided."

The team noted that Calculation BNP-MECH-E11-F028A-B, "Mechanical Analysis & Calculation for 1/2 E11-F028A/B" which was revised on March 29, 2010, calculated the minimum motor torque values using MCC minimum steady state criteria voltages instead of the more limiting MCC transient voltages for MOVs 1/2-E11-F0028A&B; and, that the licensee failed to provide a justification as required by Procedure EGR-NGGC-0101. The team also noted that the licensee had a draft "Evaluation of Electrical Transients," (developed circa 2006) that determined which MOV's were subject to electrical bus transients and evaluated the impacts/consequences of these transients on the affected MOVs.

Based on the lack of a justification for using steady-state voltages in the approved calculation, and the unapproved evaluation that assessed transient voltages, the licensee revised the draft “Evaluation of Electrical Transients,” and issued it as Attachment 4, “Evaluation of Electrical Transients” to Calculation 0BNP-TR-006, “MOV Design Basis Info – GL89-10 & GL96-05, Rev 3. The revision affected 90 MOVs that actuate during bus transients or are already stroking when a transient occurs. As a result of the revision, many of these MOVs lost available margin because use of the transient MCC voltages resulted in less available motor output torque. For example, margin was reduced from 303 ft-lbs to 4 ft-lbs for valve MOV 2-E11-F016A; and margin was reduced from165 ft-lbs to 9 ft-lbs for valve MOV 2-E11-F028B. For valve MOV-2-E11-F028B, it was necessary for the licensee to use as-tested stem factors to recover from a margin of -10 ft-lbs.

Additionally, the licensee failed to evaluate whether the 90 MOVs had adequate control power voltage to actuate the MOVs during the period they are required to operate during a design bases event. Calculation, BNP-E-1.012, “Safety-Related AC Control Loop Voltage Analysis,” Rev. 6 used steady-state post-event MCC voltages to evaluate the control circuit voltage drop for the MOVs. The use of steady-state voltages instead of transient voltages predicted higher control circuit voltages than would actually exist. The licensee reviewed the issue during the inspection and determined that 14 MOVs would not have adequate control voltage to actuate the MOVs at the receipt of their starting signal. Inadequate control voltage would result in the operation of the MOVs being delayed until the motor starting transient is completed and steady state voltages are restored at the MCCs. The licensee performed an evaluation and determined that the delayed stroking of these valves would not adversely impact the mitigation of design bases accidents. The licensee initiated nuclear condition reports (NCR) 427745 and 429541 to address these issues.

<u>Analysis</u>: The team determined that that the licensee’s failure to comply with the provisions of Nuclear Generation Group Standard Procedure EGR-NGGC-0203, “MOV Performance Prediction, Actuator Settings, & Diagnostic Test Data Reconciliation,” Rev 15, which resulted in inadequate evaluation of MOV motor actuator output torque; and the failure to evaluate whether those MOVs would have adequate control power voltage was a performance deficiency. The performance deficiency was more than minor because it was similar to Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports,” Appendix E, “Examples of Minor Issues”, Example 3j, which states that if “*the engineering calculation error resulted in a condition where there was a reasonable doubt on the operability of a system*” the performance deficiency is not minor; and because the performance deficiency adversely affected the mitigating systems cornerstone attribute of design control and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the use of non-conservative voltage inputs for MOV calculations could result in the failure of the components to perform their design bases functions during an event. In accordance with NRC IMC 0609.04,“Initial Screening and Characterization of Findings”, the inspectors conducted a Phase 1 Significance Determination Process (SDP) screening and determined the finding to be of very low safety significance (Green) because it was a design deficiency confirmed not to result in the loss of operability or functionality, did not represent the loss of a system safety function and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. By the end of the inspection, the licensee was able to demonstrate operability of the MOVs through additional analyses.

The finding involved the cross-cutting aspect of procedural compliance and personnel follow procedures within the Work Practices component of the Human Resources area because the licensee's staff failed to follow procedural guidance for using transient voltages instead of steady state voltages. [H.4(b)]

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control" requires, in part, that design control measures provide for verifying or checking the adequacy of design. Contrary to the above, the licensee failed to verify that design control measures were properly implemented when evaluating the adequacy of electrical power for MOV operation during design bases events. Specifically, the licensee failed to verify that transient voltages were analyzed as required by Procedures EGR-NGGC-0203, and EGR-NGGC-0101 during the revision of Calculation BNP-MECH-E11-F028A-B that was approved on March 29, 2010. Because this finding is of very low safety significance and because it was entered into the licensee's corrective action program as NCRs 427745 and 429541, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy: NCV 05000325/2010008-01 and 05000324/2010008-01, "Failure to Analyze MOV Operation with Transient Voltages."

.2.6 Low Service Water Pressure Switch (1-SW-PS-129)

a. Inspection Scope

The team reviewed Units 1 and 2 system flow diagrams, I&C drawings, elementary and schematic diagrams, instrument setpoint and uncertainty calculations, as well as calibration procedures and calibration test records to verify that the instrumentation and controls for the low service water pressure switch was in accordance with design bases documents. The last two completed calibration test records were reviewed to confirm that instrument setpoints were consistent with setpoint calculations. The team reviewed documentation of completed surveillance tests to verify that equipment performance was appropriately monitored and maintained consistent with the design and licensing bases. The team performed field walkdowns of the switches to observe the existing conditions. In addition, the team reviewed a sample of condition reports to confirm that the licensee adequately identified and corrected adverse conditions.

b. Findings

No findings were identified.

.2.7 4.16 kV Bus E1 Cross Tie Breaker (AG0)

a. Inspection Scope

The team reviewed the system DBD, related design bases support documentation, and operational requirements to identify the design bases requirements of the 4.16 kV bus E1 cross tie breaker. The team reviewed applicable plant operating procedures to ensure that risk significant functions could be performed. Coordination and short circuit calculations were reviewed along with maintenance and testing procedures to verify that design bases and design assumptions had been appropriately translated into calculations and procedures. The team reviewed periodic maintenance and testing practices to ensure the equipment was maintained in accordance with industry practices. System health reports, component maintenance history and licensee corrective action program reports were reviewed to verify that potential degradation was monitored or prevented and that component replacement was consistent with in service/equipment qualification life. The team also performed a visual non-intrusive inspection to assess installation, configuration, observable material condition, and potential vulnerability to hazards.

b. Findings

No findings were identified.

.2.8 Unit Substation Bus E7 K Line Breakers

a. Inspection Scope

The team reviewed the system DBD, related design bases support documentation, and operational requirements to identify the design bases requirements of the K-Line breakers associated with the 480V unit substation bus E7. Coordination and short circuit calculations were reviewed along with maintenance and testing procedures to verify that design bases and design assumptions had been appropriately translated into calculations and procedures. The team reviewed periodic maintenance and testing practices to ensure the equipment was maintained in accordance with industry practices. The associated 125Vdc voltage calculations were reviewed to verify that adequate voltage would be available for the breaker open/close coils and spring charging motors. System health reports, component maintenance history and licensee corrective action program reports were reviewed to verify that potential degradation was monitored or prevented and that component replacement was consistent with service/equipment qualification life. The team performed a visual non-intrusive inspection of observable portions of the K-Line breakers to assess installation, configuration, observable material condition, and potential vulnerability to hazards.

b. Findings

No findings were identified.

2.9 2C RHR Service Water (SW) Booster Pump

a. Inspection Scope

The team reviewed the FSAR, TS, DBD, and plant drawings to identify the design bases requirements of the 2C RHR SW booster pump. The team examined system health reports, records of surveillance testing and maintenance activities, and applicable corrective actions to verify that potential degradation was being monitored, prevented and/or corrected. After discussion with the SW and RHR system engineers, the team performed a walkdown of the pump area to examine the visible material condition of the pump. The team reviewed hydraulic calculations to verify that the potential pump degradation assumed in the IST surveillances would not prevent the pump from performing its safety-related function and that design flow and pressure requirements were correctly translated into IST acceptance criteria. The team reviewed documentation that identified features of the system differential pressure flow elements, to ensure the original design requirements had been properly incorporated into plant calibration procedures. The team reviewed calculations that establish voltage drop, protection and coordination, motor BHP requirements, and short circuit for the motor power supply and feeder cable to verify that adequate voltage was available to operate the motor.

b. Findings

No findings were identified.

.2.10 2C RHR Pump Motor Breaker

a. Inspection Scope:

The team reviewed the system DBD, related design bases support documentation, and operational requirements to identify the design bases requirements for the 4kV breakers associated with the 2C RHR pump motor breaker. Coordination and short circuit calculations were reviewed along with maintenance and testing procedures to verify that design bases and design assumptions had been appropriately translated into design calculations and procedures. The team reviewed periodic maintenance and testing practices to ensure the equipment is maintained in accordance with industry practices. The associated breaker closure and opening control logic diagrams and the 125Vdc voltage calculations were reviewed to verify that adequate voltage would be available for the breaker open/close coils and spring charging motors under accident/event conditions. System health reports, component maintenance history and licensee corrective action program reports were reviewed to verify that potential degradation was monitored or prevented and the component replacement was consistent with in service/equipment qualification life. The team performed a visual non intrusive inspection of observable portions of the pump's motor breaker to assess installation, configuration, observable material condition, and potential vulnerability to hazards.

b. Findings

No findings were identified.

.2.11 Core Spray (CS) Pump 2A

a. Inspection Scope

The team reviewed the FSAR, TS, applicable plant calculations, and drawings to identify the design bases requirements of the 2A CS pump. The team also reviewed operating procedures to verify correct implementation of design bases. The team examined vendor documentation, system health reports, records of surveillance testing and maintenance activities, and applicable corrective actions to verify that potential degradation was being monitored and prevented or corrected. The team inspected Unit 2 CS rooms to examine internal flooding potential. The team reviewed hydraulic calculations to verify that the pump degradation assumed in the IST surveillances would not prevent the pump from performing its safety-related function and that design flow and pressure requirements were correctly translated into IST acceptance criteria. The team reviewed the pump's NPSH design and vortexing calculations for both suppression pool and condensate storage tank water supplies to verify the pumps would have adequate suction head and not ingest air under accident conditions. The team also performed a walkdown to assessvisible material condition and verify that the installation and system configuration was consistent with design documentation. The team also performed interviews with plant personnel to discuss the condition of the pump. The team reviewed calculations that establish voltage drop, protection and coordination, motor BHP requirements, and short circuit for the motor power supply and feeder cable to verify that adequate voltage was available to operate the motor.

b. Findings

No findings were identified.

.2.12 4kV Bus E3

a. Inspection Scope

The team inspected the 4kV bus to verify it would operate during design basis events. System health reports, component maintenance history and licensee corrective action program reports were reviewed to verify that potential degradation was monitored or prevented. The team reviewed selected calculations for electrical distribution system load flow/voltage drop, degraded voltage protection, short-circuit, and electrical protection and coordination. This review was conducted to assess the adequacy and appropriateness of design assumptions, and to verify that bus capacity was not exceeded and bus voltages remained above minimum acceptable values under design bases conditions. Additionally, the switchgear's protective device settings and breaker ratings were reviewed to ensure that selective coordination was adequate for protection of connected equipment during worst-case, short-circuit conditions. The team evaluated selected portions of the licensee response to NRC GL 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," dated February 1, 2006. The station's interface and coordination with the transmission system operator for plant voltage requirements and notification setpoints were reviewed. The team verified the adequacy of the degraded and loss of voltage relay protection schemes and bus transfer schemes between offsite power supplies and the associated emergency diesel generators. To determine if breakers were maintained in accordance with industry and vendor recommendations, the team reviewed the preventive maintenance inspection and testing procedures. The team reviewed selected industry OE and associated plant actions to ensure that applicable insights from OE have been applied. The 125Vdc voltage calculations were reviewed to determine if adequate voltage would be available for the breaker open/close coils and spring charging motors during design bases events. And, the team performed a visual non-intrusive inspection of 4kV bus E3 to assess the installation configuration, material condition, and potential vulnerability to hazards.

b. Findings

No findings were identified.

.2.13 Unit 2 Torus Purge Exhaust Valve, 2CAC-V7-AO

a. Inspection Scope

For the wet well (Torus) vent valves, the team reviewed testing documentation involving the containment isolation override switches used to initiate the isolation override circuitry described in the plant's FSAR. The team also reviewed the use of these switches in the emergency operating procedures (EOPs) and monitoring under the maintenance rule, 10 CFR 50.65. The team also reviewed the sizing assumptions for the backup nitrogen system, as well as the accumulator, and its ability to support the component function of reopening wet well (Torus) vent valve 2CAC-V7-AO following containment isolation.

b. Findings

Introduction: The team identified an NRC identified Apparent Violation (AV) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to assure that conditions adverse to quality, such as deficiencies, were promptly identified and corrected. Specifically, after identifying that the Unit 1 and 2 isolation override switches associated with the hardened wet well vents should have been scoped in the maintenance rule, the licensee failed to ensure the circuitry was monitored for functionality. Because the circuitry was not adequately monitored, a relay in the Unit 1 circuitry degraded unacceptably without the licensee's knowledge causing the valve to not be available to perform its intended function. This finding does not present an immediate safety concern because the failed relay in the Unit 1 control circuitry was replaced and the valve's functionality restored.

Description: On November 15, 2005, the licensee initiated action request (AR) 176181 in response to an NRC inspection which identified that some components explicitly used in the emergency operating procedures (EOPs) did not have measures in place to monitor for effective maintenance as required by the maintenance rule. In response to this condition adverse to quality, the licensee initiated corrective actions including, but not limited to: 1) created a list of components explicitly identified in EOPs, 2) identified preventive maintenance and testing activities for those components, and 3) determined required action for components explicitly identified in EOPs without adequate preventive maintenance and testing activities.

The licensee's corrective actions for AR 176181 incorrectly determined that containment atmospheric control (CAC) isolation override switches and the associated control circuitry for valves 1(2)CAC-V7-AO were being monitored by maintenance surveillance tests 1(2)MST-RPS26Q, RPS High Drywell Pressure Trip Unit Channel Calibration and 1(2)MST-RPS26R, RPS High Drywell Pressure Instrumentation Channel Calibration. The existing control switches were utilized during the tests. However, the switches are a three position switch (off-neutral-override) and the override position was never selected during the test procedure. As a result, the isolation override circuitry, including its relays, were not being functionally tested as they would be used in the EOPs nor were they monitored for reliability. The isolation override control switches allow the 1(2)CAC-V7-AO valves to be reopened following a containment isolation signal by manually overriding the automatic closure signal, allowing for post loss of coolant accident (LOCA) venting of the suppression pool airspace via the hardened wet well vent. The vent's basic function is for venting excess containment pressure resulting from steam generation in the suppression pool during a loss of decay heat removal sequence. The path provides a method to prevent exceeding the primary containment pressure limit of 70 psig. After venting, the probabilistic safety analysis credits the use of the low pressure coolant injection (LPCI) and core spray (CS) systems.

The failure to identify and correct this condition adverse to quality was entered into the licensee's corrective action program as nuclear condition report (NCR) 428054, "CAC Override Switches Function Not Tested in Override." The licensee performed an extent of condition review that identified 4 isolation override switches associated with 14 valves on each unit that required testing. The testing in January 2011 revealed the failure of a Unit 1 relay (3-12) which adversely affected the override capability associated with Unit 1 valve CAC-V7-AO.

Analysis: The licensee's failure to comply with licensee initiated corrective actions to establish adequate maintenance surveillance tests to ensure that the isolation override control switches were being tested in the manner that they would be operated in the EOPs, which resulted in the override capability of safety-related valves being adversely affected, was a performance deficiency. The performance deficiency was more than minor because it is associated with the mitigating systems cornerstone attribute of equipment performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, not testing the isolation override circuitry resulted in a failed Unit 1 component going undetected and adversely impacted the ability to mitigate an event with the hardened wet well (Torus) vent. Using Manual Chapter Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the inspectors determined the finding required a Phase 2 analysis because the finding represented an actual loss of safety function of non-Technical Specification equipment designated as risk significant for greater than 24 hours. The Phase 2 analysis determined that this finding was potentially greater than green; therefore, a Phase 3 analysis was performed by a regional senior reactor analyst (SRA). The SRA determined the change in risk through use of the plant specific risk model. The function of the wet well vents was modeled, but the model did not include the ability to supply air manually to the air operated valve as a backup to the control room switches. A human reliability analysis was performed, and model adjustments were made so the performance deficiency's impact could be analyzed, given the backup method was available. This backup method's availability resulted in the findings risk increase to be low enough to be considered a Green SDP item. A cross-cutting aspect was not identified because the finding does not represent current performance.

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," required in part, that conditions adverse to quality, such as deficiencies, are promptly identified and corrected. Contrary to the above, a condition adverse to quality was not promptly identified and corrected. Specifically, on October 19, 2010, it was identified that the licensee had not taken appropriate corrective actions in response to AR 176181, written on November 15, 2005, which identified that some components used in the EOPs were not being tested or otherwise monitored for effectiveness of maintenance. Specifically, the licensee failed to identify that the Units 1 and 2 isolation override control switches and associated circuitry for CAC-V7-AO valves were not being functionally tested.

Subsequent testing in January 2011 determined that the override capability of the Unit 1 valve was adversely affected by a failed relay. Because this finding is of very low safety significance and because it was entered into the licensee's corrective action program as NCR 428054, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy: NCV 05000325/2010008-02 and 05000324/2010008-02, "Failure to Promptly Identify and Correct Isolation Override Circuitry Testing Deficiencies."

.2.14 RHR Pump 2C

a. Inspection Scope

The team reviewed the FSAR, TS, DBD, System Description, applicable plant calculations, and drawings to identify the design bases requirements of the 2C RHR/Low Pressure Coolant Injection (LPCI) pump. The team examined system health reports, records of surveillance testing and maintenance activities, and applicable corrective actions to verify that potential degradation was being monitored, prevented and/or corrected. After discussion with the system engineers, the team performed a walkdown of the RHR Loop 2A area to examine the visible material condition of the pump, and to verify that the installation was consistent with design documentation. The team reviewed hydraulic calculations to verify that the potential pump degradation assumed in the IST surveillances would not prevent the RHR pump from performing its safety-related function and that design flow and pressure requirements were correctly translated into IST acceptance criteria. The team reviewed documentation that identified features of the system differential pressure flow elements, to ensure that the original design requirements had been properly incorporated into plant calibration procedures. The team also reviewed the hydraulic calculations to verify that runout flow, NPSH and vortex calculations for the pump were acceptable, and in conformance with pump vendor requirements. The team reviewed documentation, calculations, operating procedure cautions, and testing results to ensure that pump minimum flow concerns identified in NRC Bulletin 88-04, Potential Safety-Related Pump Loss had been adequately addressed. The team reviewed calculations that establish voltage drop, protection and coordination, motor BHP requirements, and short circuit for the motor power supply and feeder cable to verify that adequate voltage was available to operate the motor.

b. Findings

No findings were identified.

.2.15 RCIC Turbine Exhaust Vacuum Breaker Valves (E51-F063 and E51-F064)

a. Inspection Scope

The team reviewed the RCIC turbine exhaust vacuum breaker valves, E51-F063 and E51-F064, to verify that they were capable of performing their design basis requirements. The team reviewed test procedures involving the valves to verify the ability of the valves to perform their design function was being adequately demonstrated. Modifications to the valves and adjacent piping were also reviewed to ensure that modifications did not have a negative impact on the performance of the valves. Additionally, the team reviewed the timing of the RCIC turbine trip valves relative to the timing of the exhaust vacuum breaker line containment isolation valves, E51-F062 and E51-F066, to verify the sequence of valve closure did not negatively impact the ability of the vacuum breakers to prevent waterhammer events following RCIC turbine isolation.

b. Findings

No findings were identified.

.2.16 RCIC CST Suction Valves (1(2)-E51-F010)

a. Inspection Scope

The team reviewed the Units 1 and 2 RCIC CST Suction Valves, E51-F010 to verify they were capable of performing their design basis function. The team reviewed the licensee's calculations of operational margin and verified important inputs into the calculations were sufficiently conservative. The team also verified the in-field setup of switch settings for the valve actuators were within the setup window assumed in design margin calculations, and verified that test equipment accuracies were considered. The team verified appropriate assumptions were used relative to the direct current (DC) energy source for the motor operators such that the worst case voltage was used as an assumption for available motor output torque when determining margin. IST of the valves was reviewed to verify the performance of the valves was being trended such that degradation of the valves would be identified before performance requirements were exceeded.

b. Findings

No findings were identified.

.2.17 EDG Auxiliaries - Air Start System

a. Inspection Scope

The team reviewed the FSAR, TS, applicable plant calculations, and drawings to identify the design bases requirements and operational requirements for the emergency diesel generator air start system. Calculations supporting the installed system capability were reviewed to verify that design bases and assumptions were appropriately translated and that conclusions supported overall system capability. The team performed a field walkdown to assess visible material condition and verify that the installation and system configuration was consistent with design bases and plant drawings. Control panel indicators were observed and operating procedures reviewed to verify that component operation and alignments were consistent with design and licensing basis assumptions.

b. Findings

No findings were identified.

.3 Review of Low Margin Operator Actions

a. Inspection Scope

The team performed a margin assessment and detailed review of the following three risk-significant and time-critical operator actions. Where possible, margins were determined by the review of the assumed design basis and UFSAR response times, and performance times documented by job performance measures (JPMs). The team performed an assessment of the EOPs, abnormal operating procedures (AOPs), annunciator response procedures (ARPs), and other operations procedures to determine the adequacy of the procedures and availability of equipment required to complete the actions. Operator actions were observed on the plant simulator and during plant walk downs as appropriate.

- Operator Action to crosstie Conventional Service Water (CSW) to nuclear service water (NSW) on loss of offsite power (LOOP)
- Operator Action to initiate control rod drive (CRD) pump injection due to late injection of RCIC and high pressure coolant injection (HPCI)
- Operator Action to align the condensate system for low pressure injection

b. Findings

Introduction: An NRC-identified Green non-cited violation (NCV) of Technical Specification (TS) 5.4.1, Procedures, was identified for failure to maintain adequate abnormal operating procedures (AOP) for opening a service water cross tie valve during a loss of offsite power (LOOP) event. The valve would not open against system differential pressure (dp) and the licensee's corrective actions did not address the valve's manipulation in AOPs.

Description: Conventional Service Water (CSW) supply to the reactor building closed cooling water (RBCCW) is provided through crosstie valve 1(2)-SW-V146 which is a manual valve that is opened during a LOOP or loss of nuclear service water (NSW) event to supply cooling to RBCCW heat exchangers (HX). The HXs supply cooling water to the drywell coolers, which maintain the drywell less than 150°F as required by TS. The RBCCW HXs also supply cooling water to control rod drive (CRD) pumps, and the reactor water cleanup system. Additionally, the crosstie valve is used to provide an alternate supply to the residual heat removal HXs.

The team reviewed nuclear condition report (NCR) 245829 which documented an event when two auxiliary operators could not get service water valve 1-SW-V146 open during a maintenance evolution. The licensee assigned two corrective actions to address the deficiencies with valve 1-SW-V146. The first corrective action was to perform maintenance on the valve; however, this corrective action was cancelled because the second shift of operators was able to successfully open the valve under a similar plant configuration. The second corrective action addressed changes to the maintenance procedure to ensure another valve remains open until 1(2)-SW-V146 is open to allow minimal dp across valve 1(2)-SW-V-146. This corrective action did not address the requirements to open the valve in AOP-18.0, Nuclear Service Water Failure, Rev 21 and AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses, Rev 52. Consequently, the licensee may not be able to open valves 1(2)-SW-V146 with maximum dp during a LOOP event. Therefore, the team determined that inadequate guidance existed in the AOPs to ensure the operator action could be completed. This finding has been entered into the licensee's corrective action program (CAP) as NCR 00428809.

<u>Analysis</u>: The licensee's failure to provide appropriate procedural guidance as required by plant Technical Specifications to assure AOP-18.0 and AOP-36.1 could be performed, which resulted in the potential inability to operate valves 1(2)-SW-V146 during a LOOP event was a performance deficiency. The performance deficiency was more that minor because it is associated with the mitigating systems cornerstone attribute of equipment performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences and, if this condition was left uncorrected the ability to complete required operator actions in AOP18.0 and AOP36.1 during a LOOP would be adversely affected. Using Manual Chapter Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the inspectors determined the finding required a Phase 2 analysis because the finding represented an actual loss of safety function of non-Technical Specification equipment designated as risk significant for greater than 24 hours. The Phase 2 analysis determined that this finding was potentially greater than green; therefore, a Phase 3 analysis was performed by a regional senior reactor analyst (SRA). The SRA determined the combined risk associated with the valve's function to provide alternate flow to RBCCW and RHR heat exchangers was very low. In addition, there was a significant probability of operator recovery due to the long time period that was available before manipulation of the valve was required during performance of the AOPs. These factors resulted in a risk value corresponding to a Green finding. The finding directly involved the cross-cutting aspect of thoroughness of evaluation within the Corrective Action Program component of the Problem Identification and Resolution area because the licensee failed to properly evaluate the inability to open valve 1(2)-SW-V146 when the issue was entered into the CAP [P.1(c)].

<u>Enforcement</u>: TS 5.4.1 states, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, November 1972 (Safety Guide 33, November 72). Section F of Safety Guide 33 recommends procedures for "Combating Emergencies and Other Significant Events." Contrary to the above, the licensee failed to maintain adequate procedures for combating a significant event. Specifically, in September 2007, the licensee failed to provide appropriate procedural guidance in procedures AOP-18.0, Rev 21 and AOP-36.1, Rev 52 to assure the operator's ability to open valve 1-SW-V146 against maximum dp during a LOOP event. Because this finding is of very low safety significance and because it was entered into the licensee's corrective action program as NCR 428809, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy: NCV 05000424/2010008-03 and 05000425/2010008-03, "Failure to Perform Appropriate Corrective Actions for Deficiencies with Opening a Service Water Valve."

.4 Review of Industry Operating Experience

a. Inspection Scope

The team reviewed selected operating experience issues that had occurred at domestic and foreign nuclear facilities for applicability at the Brunswick Steam Electric Plant. The team performed an independent applicability review for issues that were identified as applicable to the BNP and were selected for a detailed review. The issues that received a detailed review by the team included:

- IN 2008-18: Loss of a Safety-Related Motor Control Center Caused by a Bus Fault
- IN 2009-04, Age-Related Constant Support Degradation
- IN 2007-01, Recent Operating Experience Concerning Hydrostatic Barriers
- 10 CFR Part 21 Reports and Operating Experience Relating to Tyco/Agastat E7000 Relays

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA6 Meetings, Including Exit

On February 3, 2011, the team presented the inspection results to Mr. Annacone and other members of the licensee’s staff. Proprietary information that was reviewed during the inspection was returned to the licensee.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
A. Pope, Licensing
T. Sherrill, Licensing
J. Titrington, Manager, Design Engineering

NRC personnel
B. Desai, Chief, Engineering Branch Chief 1, Division of Reactor Safety, RII

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000325 & 324/2010008-01	NCV	Failure to Analyze MOV Operation with Transient Voltages (Section 1R21.2.5)
05000325 & 324/2010008-02	NCV	Failure to Promptly Identify and Correct Isolation Override Circuitry Testing Deficiencies (Section 1R21.2.13)
05000325 & 324/2010008-03	NCV	Failure to Perform Appropriate Corrective Actions for Deficiencies with Opening a Service Water Valve (Section 1R21.3)

LIST OF DOCUMENTS REVIEWED

Calculations

04KV-0002, 4.16kV Emergency Bus Degraded Grid Voltage Relay Setpoint Calculation, Rev. 1
0B11-0026, Core Spray Allowable Leakage, Rev 5
0B21-0098, OPL-4 and OPL-5 SAFER/GESTR-LOCA Analysis Input, Rev. 2
0B21-0199, ECCS Analysis Results, Rev. 6
0BNP-TR-006, MOV Design Basis Information – GL89-10 & GL 96-05, Rev. 3
0DSA-001, Determine if Receiver Pressure After 12 Seconds of Jet Assist Can Provide Control Air, Rev. 0
0DSA-005, Diesel Generator Starting Air Requirements, Rev. 0
0E11-0023, Power Uprate Minimum Flow Rate Uncertainty and Scaling Calculation, Rev. 2
0E11-028, Determination of RHR and Core Spray NPSH Margins After Power Uprate, Rev. 5
0E21-0002, Determination of Core Spray NPSH margins, Aligned to CST, Rev. 0
0E41-1001, HPCI and Condensate Storage Tank Level-Low Uncertainty and Scaling Calculation (E41-LSL-N002(3) Loops), Rev. 2
0E51-0028, RCIC and Condensate Storage Tank Level-Low Uncertainty and Scaling Calculation (E51-LSL-4463(4) Loops), Rev. 2
0EOP-WS-13.1, LPCI/RHR Vortex Limit (2 Pumps) Plus HPCI & RCIC Vortex Determination, Rev. 5
0EOP-WS-13.3, Core Spray Vortex Limit, Rev. 4
0EOP-WS-19, Core Spray Pump NPSH Limit, Rev. 5
80-004-02, RHR Pumps Minimum Flow Bypass Line Flow, Rev. 2
95135-C-24, Seismic Weak Link Assessment MOVs:½-E11-F028 A/B & ½-E11-F047A/B, Rev 3
ANP-2624(P), Brunswick Units 1 and 2 LOCA-ECCS Analysis MAPLHGR Limit for Atrium -10 Fuel, Rev. 2
BNP-E-1.012, Safety-Related AC Control Loop Voltage Analysis, Rev. 6
BNP-E-2.002, 480V AC Safety-Related MOV Electrical Protection, Rev. 8
BNP-E-2.007, U2 480V Vital MCC Calculations, Rev. 6
BNP-E-7.002, AC Auxiliary Elect. Distribution Sys. for Voltage/Load Flow/Fault Current Study, Rev. 5H
BNP-E-7.010, Emergency Diesel Generator Static & Dynamic Load Study, Rev. 7
BNP-E-8.010, AC Coordination Study, Rev. 9
BNP-E-8.013, Motor Torque Analysis for AC Motor Operated Valves Unit 1, Rev. 7
BNP-E-8.013, U1 Motor Torque Analysis for AC MOVs, Rev. 7
BNP-E-8.014, U2 Motor Torque Analysis for AC MOVs, Rev. 7
BNP-MECH-E11-F009, Mechanical Analysis and Calculations for 1 & 2-E11-F009 Shutdown Cooling Inboard Suction Isolation Valves, Rev. 5
BNP-MECH-E11-F028A/B, Mechanical Analysis and Calculations for 1 & 2 E11-F028A/B RHR Suppression Pool Discharge Isolation Valve, Rev. 5
BNP-MECH-E51-F010, Mechanical Analysis and Calculations for 1-E51-F010 & 2-E51-F010 RCIC Condensate Storage Tank Suction Valves, Rev. 3
BNP-MECH-MOV-PL, Review of BNP Test Data to establish bounding packing loads for BNP GL 89-10 Motor-Operated Valves, Rev. 2
BNP-MECH-MOV-ROL, Review of BNP “As Tested” ROL Data and Determination of ROL Values to Be Used for BNP GL 89-10 Motor-Operated Valves, Rev. 3
BNP-MECH-MOV-SF, Review of BNP Test to Establish Bounding Stem Factors for BNP GL 89-10 Motor-Operated Valves, Rev. 4
EC63657, Attachment B - Hydraulic Calculation for Core Spray, 5/2006

EGR-NGGC-0203, Motor Operated Valve Performance Prediction, Actuator Settings, and Diagnostic Test Data Reconciliation, Rev. 15
G0050A-12, BNP Unit 2 Service Water System Hydraulic, Rev. 7
PCI-NPD-CPL01, Head Loss Calculations for Bare Sure-Flow™ Suction Strainers at Brunswick 1 and 2 Nuclear Units, Rev. 2
PCN-G0050A-13, NSW and CSW Pressure Switch Setpoints for Pump Auto Start, Rev. 1

Completed Procedures

0PT-07.2.4a, Core Spray System Operability Test – Loop A, 12/18/08, 4/1/09, 6/6/2009, 8/27/2009, 11/19/2009, 2/11/2010, 5/7/2010, and 7/29/2010
0PT-08.11.L, LPCI/RHR System LOOP B Valves Local Control Operability Test, 8/5/2010
0PT-08.2.2b, LPCI/RHR System Operability Test – Loop B, 6/24/2010
0PT-10.1.8, RCIC System Valve Operability Test, 10/16/2009, 1/6/2010, 1/20/2010, 3/4/2010, 3/30/2010, 4/15/2010, 4/21/2010, 4/25/2010, 5/27/2010, 6/24/2010, and 8/19/2010
0PT-20.10, Testing of Valves E41-F076, E41-F077, E51-F063, and E51-F064, 3/24/2008, 4/5/2008, 11/22/2008, 11/23/2008, 3/3/2009, 3/25/2009, 4/5/2009, 3/9/2010, 4/2/2010, and 4/7/2010
0PT-20.3, Local Leak Rate Testing, 3/8/2005 and 3/7/2007
0PT-20.3-E11, Local Leak rate Testing for Residual Heat Removal System, 3/30/2008 and 3/22/2009
1PT-24.1-1, Service Water Pump and Discharge Valve Operability Test, 9/9/2010
2PT-24.1-2, Service Water Pump and Discharge Valve Operability Test, 5/1/2010, 7/17/2010, and 7/22/2010
OPT-08.1.4a, OPT-08.1.4a, RHR Service Water System Operability Test, Loop A, 5/15/2010 and 9/20/2010
OPT-08.2.2c, LPCI/RHR System Operability Test – Loop A, 2/18/2009 and 7/8/2010
OSP-96-009, Data Acquisition for LPCI/RHR System Resistance Test, 7/21/1997

Completed Work Orders

00045270, Perform 0PDM-MO005A (VOTES Testing), 6/25/2004
00542374, Cal 1-SW-PS-129 and 1- SW-PS-3213, 8/22/2005
00609081, Calibrate 1-E51-PSH-N009A, 3/16/2006
00613319, Calibrate RCIC Turbine Exhaust Pressure Switch, 5/10/2006
00986748-01, Calibrate RHR Pumps 2A/2C Discharge Flow Pressure Switch 2-E11-PDIS-N021, 2/18/2009
01086654, 1-E11-F009 Rising Stem Static Test Analysis, 4/4/2008
01114377, Calibrate 1-E51-PSH-N009A, 6/25/2009
01117559 01, 0MST-DG13R DG-3 LOOP-LOCA Load Test, 05/01/2009
01119165, Calibrate RCIC Turbine Exhaust Pressure Switch, 5/27/2009
01297454, 2-E11-F009 Replace the Operator, 9/24/2008
01297454, 2-E11-F009 Rising Stem Static Test Analysis, 3/22/2009
01320502, 1-E51-F064, Valve Found Sticking, 3/27/2008
01323967, Cal 1-SW-PS-129 and 1- SW-PS-3213, 8/04/2009
01324449, 1-E11-F009-MO, 0PM-MO504 Mech and Lube Inspection, 6/30/2009
01369163, Correct Pipe Slope to Eliminated High Point, 3/03/2009
01378542, 1-E51-F064: Disassembly Check Valve, 7/1/2008
01385837, 1-E51-F063 Disassemble and Clean, 9/3/2008

01385837-01, RCIC Turbine Exhaust Vacuum Breaker…Check Valve, 11/22/08
01385838, 1-E51-F064 Disassemble and Clean, 9/3/2008
01421039, 1-E51-F010 Inspect/ Repair Disc, 6/2/2009
01486171-01, Calibrate 2-E11-FT-N015A, 2E11-FY-5119A, 6/13/2010
01511085, 2-E51-F063, Failed Closure Testing, 3/5/2009
01536284, 2MST-HPCI27Q HPCI and RCIC CST Low Water Level Inst Chan, 12/08/2009
01545930, 1-E51-56-2-152 Correct Pipe Slope, 3/22/2010
01581319, 1MST-HPCI27Q HPCI and RCIC CST Low Water Level Inst Chan, 3/19/2010
01581327, 2MST-HPCI27Q HPCI and RCIC CST Low Water Level Inst Chan, 3/19/2010
01631012, 1MST-HPCI27Q HPCI and RCIC CST Low Water Level Inst Chan, 6/14/2010
01668956, 2-E11-F009-MO ESR 00-00025, Packing Adjustment Evaluation, 2/1/2010
01736798, 1-E51-F064 Closure Problem, 4/3/2010

Corrective Action Documents

00174997, RHR F028 and LOOP Pressurization Operability
00176181, MR Scoping & Maintenance on EOP Components
00190346, Adequacy of Screen 04-1255
00214171, Unplanned LCO Entry on Unit 2 SP Cooling
00217566, 2A RHR LOOP Pressurization Causes SPC Inoperability.
00220519, E1-E3 Crosstie Breaker Misaligned
00224408, Unplanned LCO Entry 2-E11-F009 Failed to Open for SDC
00234689 HPCI Discharge Check Valve Should be Tested in the Closed Direction
00245829 1B RDR SW Pump Tripped While Attempting to Align RCC to CSW
00246526, 1-E51-F063 Failed Opening Portion of 0PT-20.10
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00274009, IN 2008-18 Applicability Review
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